Addressing the Issue of PRA Quality, Regulatory Guide: *"An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities"*

Presented by Mary Drouin
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US Nuclear Regulatory Commission
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BACKGROUND/HISTORY

- PRA Policy Statement
  - Encourages staff use of PRA in all regulatory matters
- GAO
  - Indicated need to “develop standards on the scope and detail of risk assessments...”
- DSI-13
  - “...where there are needs for new codes, standards, and guides and recommendations for areas of emphasis. The NRC’s initial activities .... should include development in Probabilistic Risk Assessment (PRA)...”
- April 18, 2000, SRM
  - Indicated that the staff “should provide its recommendations to the Commission for addressing the issue of PRA quality...”

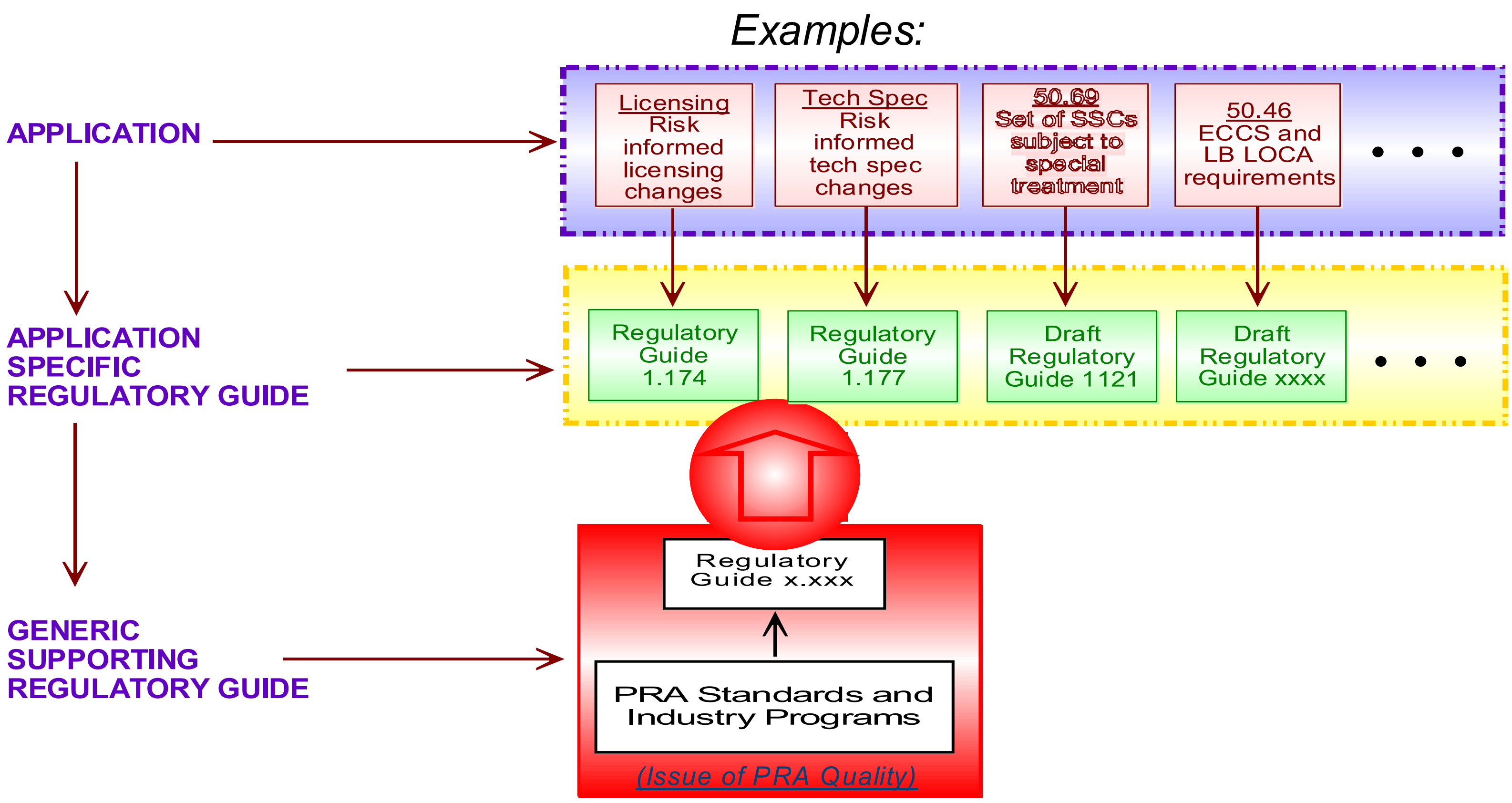
BACKGROUND/HISTORY (cont'd)

- October 27, 2000 SRM
  - Commission indicated that "the timely resolution of PRA quality requirements is necessary to support existing and developing risk-informed regulation..."
- April 5, 2002
  - ASME published ASME RA-S-2002 "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications"
- Summer 2002
  - NEI provided "Self-Assessment Process" to address differences between ASME standard and NEI 00-02
- SECY-02-0070
  - Staff indicated its plan "to develop a new RG and SRP chapter that would provide guidance to licensees and the staff, respectively, on how to use the standards and other industry programs in evaluating the technical appropriateness of PRA results for risk-informed applications"

SCOPE OF REGULATORY GUIDE

- Does not address how PRA results are used in a decision-making process
- The guidance on how PRA results are used in a risk-informed activity is addressed in the application specific regulatory guide
- This RG (and associated SRP) solely addresses the issue of determining the acceptability of the base PRA results that are used for an application

RELATIONSHIP OF GUIDE TO RISK-INFORMED ACTIVITIES



PURPOSE OF REGULATORY GUIDE

- Staff recommendation for addressing the issue of “PRA Quality” to support risk- informed regulatory activities
- Guidance to licensees and guidance to the staff
- When used in support of an application, should obviate the need for an in-depth review of the PRA by NRC staff
- Provide for a more focused and consistent review process

ORGANIZATION OF REGULATORY GUIDE

- ***<u>Main Body</u>***: provide regulatory position on the issue of “PRA Quality” to support risk-informed regulatory activities
- ***<u>Appendices</u>***: provides regulatory position on specific PRA standard or programs
  - Appendix A — NRC regulatory position on ASME PRA standard
  - Appendix B — NRC regulatory position on the NEI peer review and self-assessment process
- Future appendices — NRC regulatory position on ANS standards on external hazards, low power shutdown, internal fires, and on any other PRA standards or programs

MAIN BODY OF REGULATORY GUIDE

- Guidance provided in four areas:
  - Minimal set of functional requirements of a technically acceptable PRA
  - NRC position on consensus PRA standards and industry PRA program documents
  - Demonstration that the PRA used in regulatory applications is of sufficient technical adequacy
  - Documentation to support a regulatory application

APPENDICES

- ***<u>No objection</u>***: the staff has no objection to the requirement
- ***<u>No objection with clarification</u>***: the staff has no objection to the requirement; however, certain requirements, as written, are either unclear or ambiguous and therefore, the staff has provided their understanding of these requirements
- ***<u>No objection subject to qualification</u>***: the staff has a technical concern with the requirement and has provided the needed qualification to resolve the concern

STATUS

- November 2002
  – guide issued for public review and comment
- April 2003
  – received comments from ACRS on guide
- June 2003
  – peer review of SONGS PRA using ASME standard
- Revised guide (based on above input) and ready to issue for “trial use”

TEST REGULATORY GUIDE VIA PILOT(S)

- Purpose of trial use:
  - Determine if implementation of regulatory guide achieves its objective
- Provide assistance and clarification; for example,
  - Interpretation of documentation needs
  - Interpretation of requirements
  - Interpretation on staff position
- Provide guidance on scope and level of detail of staff review to provide consistency and uniformity in the reviews
- For pilot only, a “detailed” review may be required to identify areas of clarification, etc.
  - In form of audit

ADDITIONAL PURPOSES OF PILOT(S)

- Additional objective to gaining insights on the guide during the trial use and pilot applications
  - Identify what parts of the PRA are needed for specific applications
- Create some type of matrix that illustrates in one place what parts of the PRA are needed for different applications
  - Perhaps added to this regulatory guide
  - Can be expanded as different types of applications are added

Test of Regulatory Guide

- Does the guide provide clear and sufficient guidance to meet its objective and purpose? For example,
  - Guidance on submittal documentation?
  - Staff objections?
- Need to identify guidance and requirements that, for example,
  - there is disagreement, to test for resolution
  - are essential, to test for appropriate interpretation

Next Steps . . . .

- Publish regulatory guide for trial use
  - December 2003
- Identify pilot applications (beyond South Texas Tech Spec Initiative)
- Develop “guidance” for pilot applications
  - Draft December 2003
- Commence pilot(s) early 2004
- Modify regulatory guide (and SRP) and issue as Revision 0, late 2004 (dependent on application)
- Provide feedback to ASME and NEI
- Future pilots on future appendices (e.g., external hazards, low power shutdown, internal fire)

Experience With PRA In The Generic Issues Program

Harold J. VanderMolen
Generic Issues team
Office of Nuclear Regulatory Research

What is a Generic Issue?

- *"A regulatory matter involving the design, construction, operation, or decommissioning of several, or a class of, NRC licensees or certificate holders that is not sufficiently addressed by existing rules, guidance, or programs"*
- Must involve safety (or burden reduction)
- Must affect at least two dockets
- Must not already be covered by existing regulations & guidance

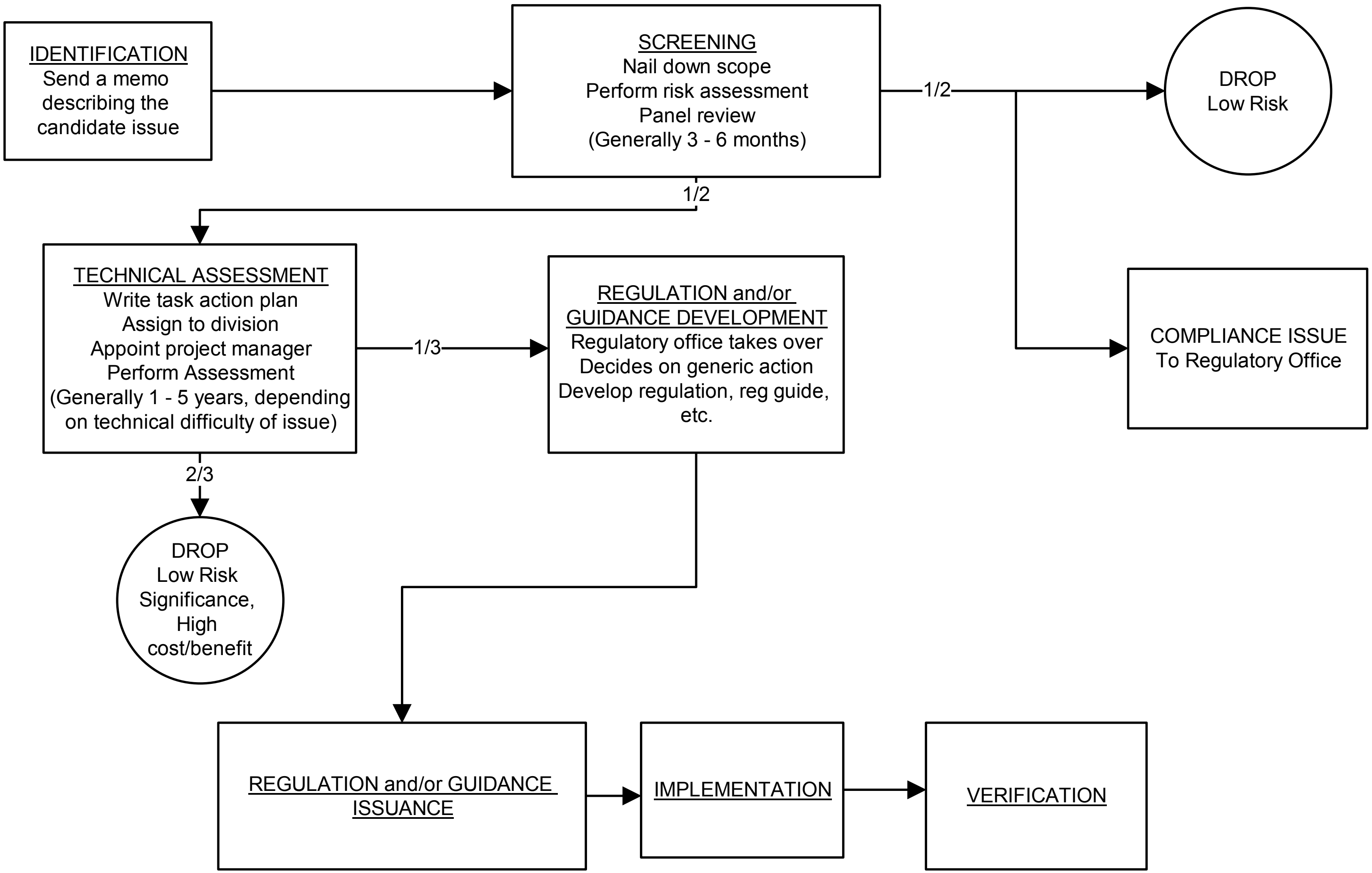
Origins of Generic Issues

- CP/OL reviews
- ACRS concerns
- TMI-2 accident
- Operating experience
- Differing Professional Opinions
- Staff concerns
- Nuclear Power Industry
- Public
- Part 21 notification
- Accident Investigations
- Event Investigations

Practical Aspects

- GI Program is not an adversarial process
- Some involvement from the issue initiator is needed
- GI program is not intended for immediate or emergency action
- GIs are often difficult – or they would be addressed by other programs
- GIs tend to be long term projects – at least a year
- After the screening stage, the end date is reported to the Congress

Generic Issue Stages



Advantages of Probabilistic Screening Analysis

- Enforces discipline in defining problem
- If an issue is dropped, there is a defensible basis
- Issues that pass are in a good position for resources
- Any disputes tend to be clearly focused

Advances in PRA Tools

- Saphire code (runs on a personal computer)
- Availability of NUREG-1150 models
- Availability of SPAR models

Nature of Calculation

- Usually want the <u>change</u> in CDF associated with the issue
- Also want error analysis, if practical
- Governed by NUREG-1489, "A Review of NRC Staff Uses of Probabilistic Risk Assessment

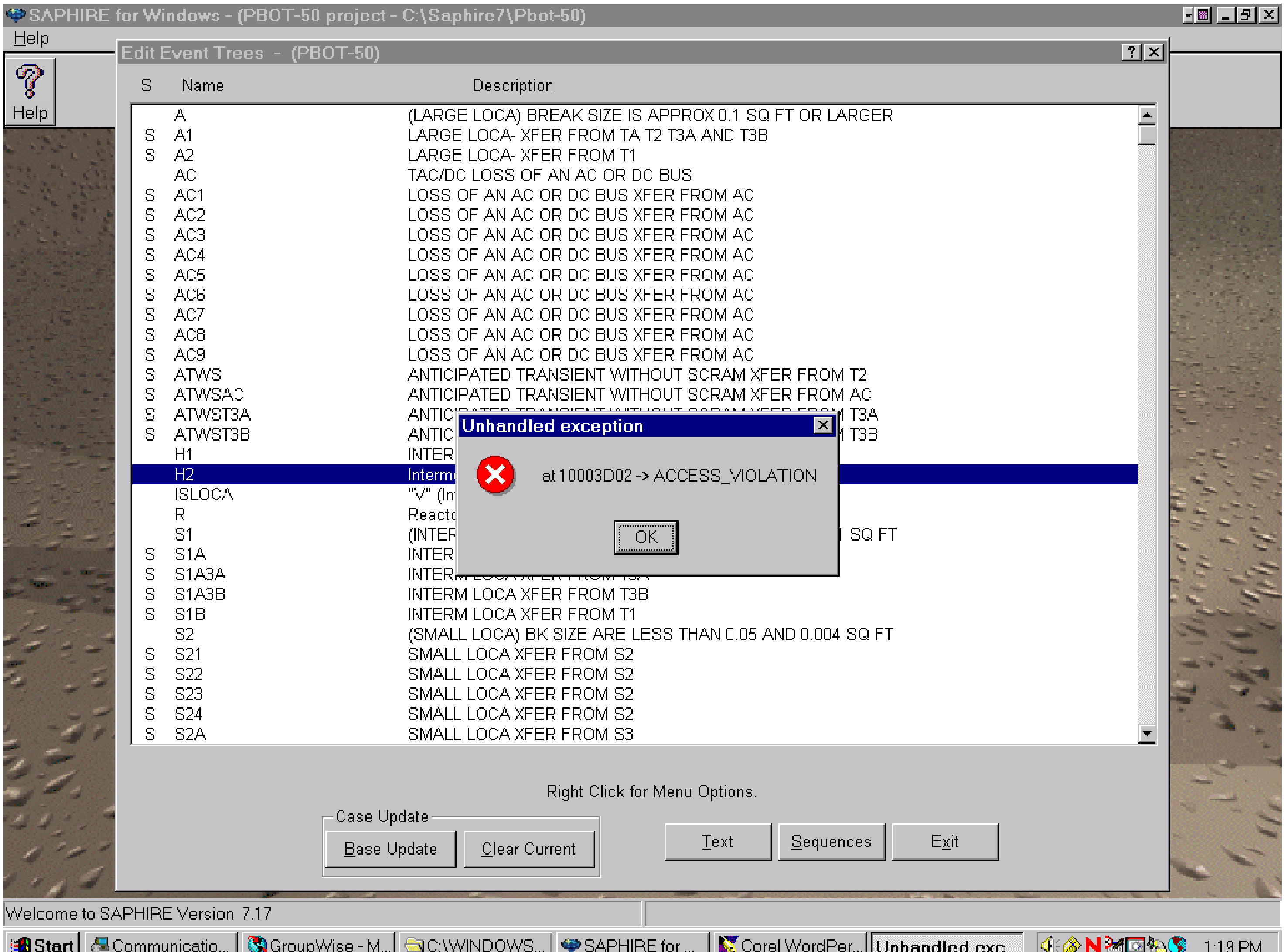
Nature of Calculation

Change in initiator frequency	• Change initiating event freq for event tree to change in IE freq • Shut off all other event trees
Change in system reliability	• Change split fraction • Shut off all other event trees
Change component reliability	• Change basic event parameter in data library • Accumulate sample runs with and without change • Form distribution of differences
New phenomenon or accident sequence	• Generate new event tree • Shut off all other event trees
New system interaction	• Link fault trees, use rules editor

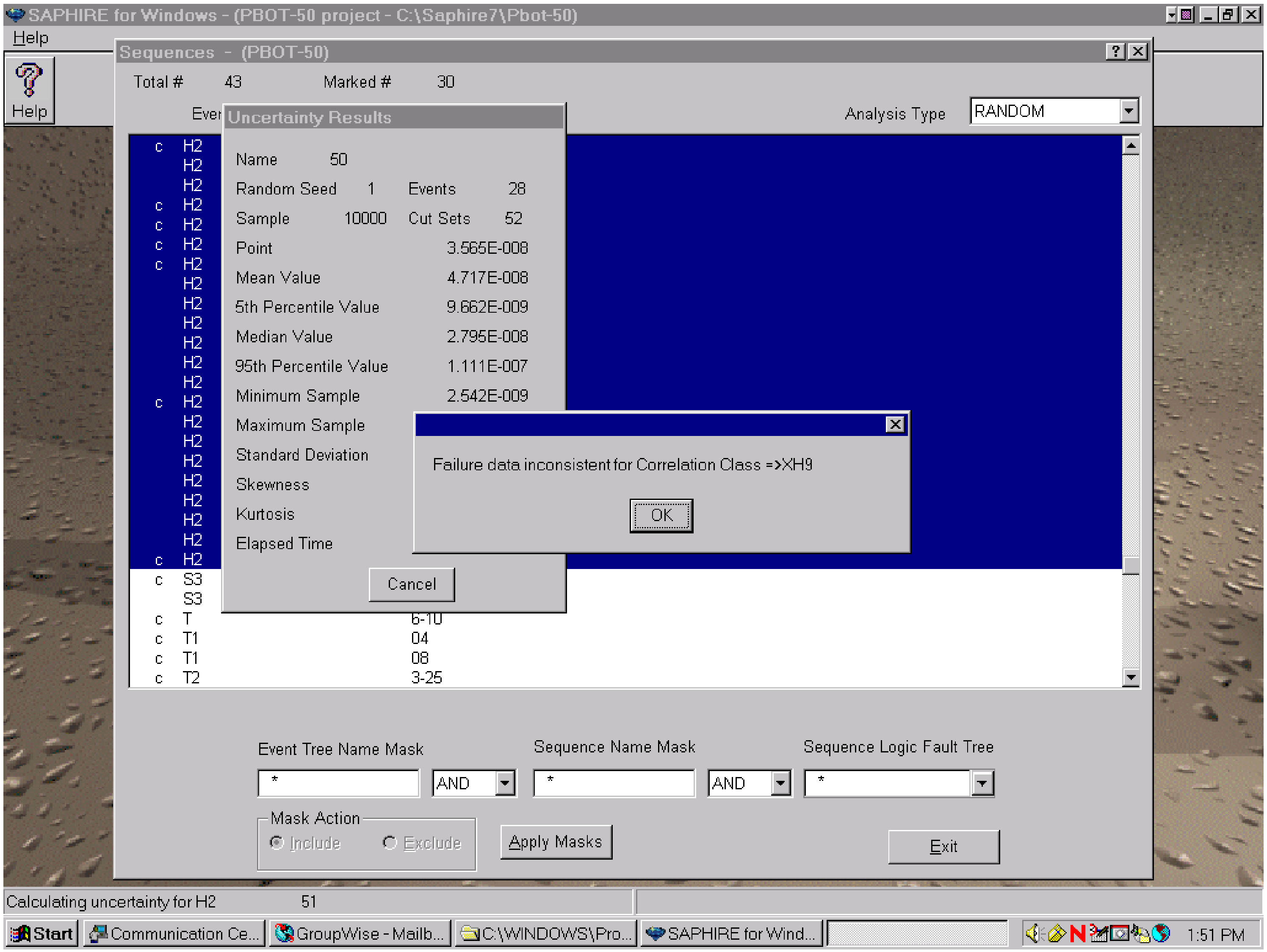
Common Problems

- Code has some idiosyncrasies
- Can discover problems with original model
- Original approximations (e.g., truncation level) may not be appropriate
- Must sometimes use some conservatism
- New accident sequences can be complex

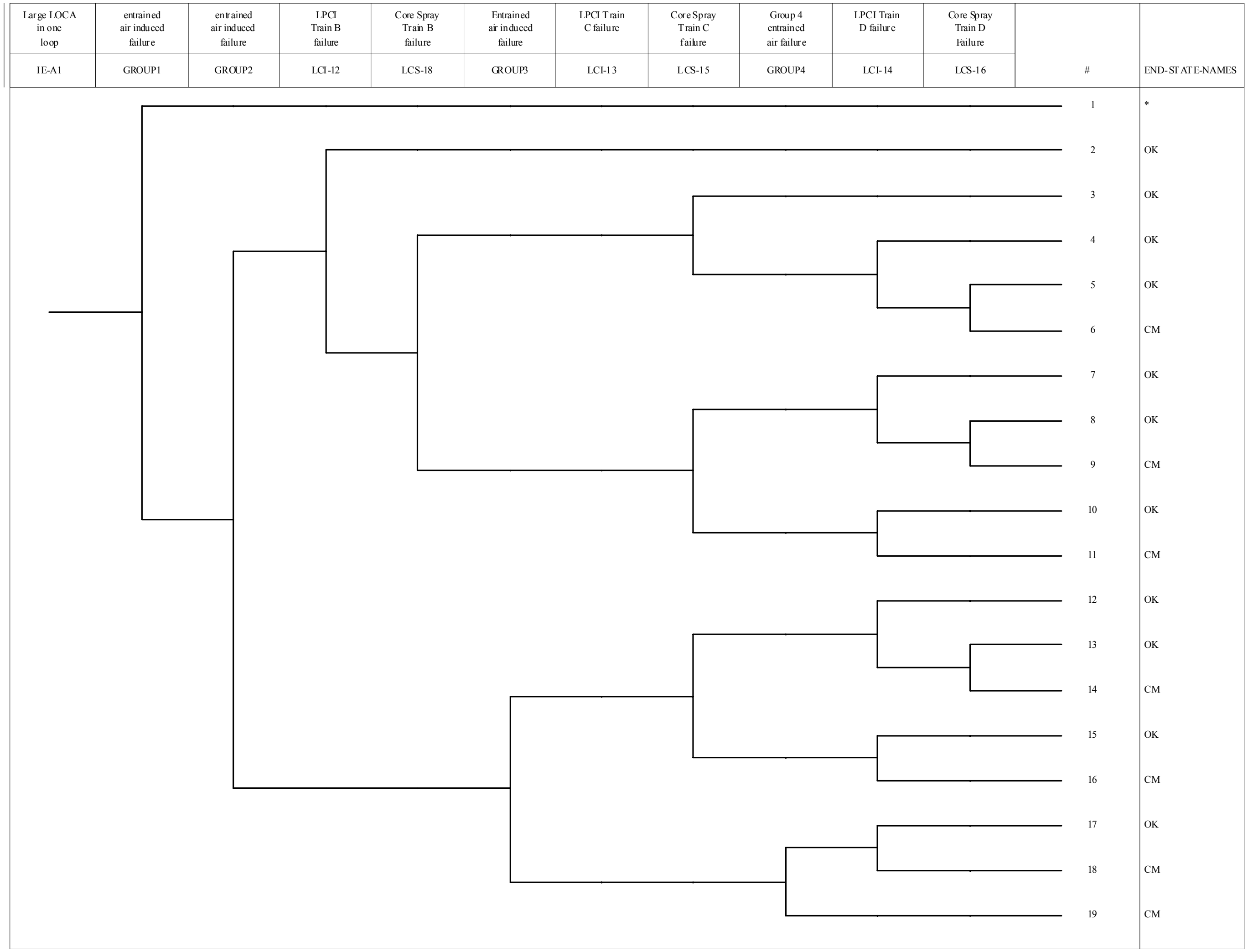
Example – code problems



Example – model problems



Example – new accident sequence



Recommendations

- Always review the top few sequences
- Try to do one sequence by hand

Recent Generic Issues

- GI-185 – Control of Recriticality following Small-Break LOCAs in PWRs
- GI-193 – BWR ECCS Suction Concerns
- GI-195 – Hydrogen Combustion

Conclusion

Boredom never seems to be a problem

Preliminary Results from a Pilot Application of a Risk-Informed Approach for Certification of Spent Fuel Storage Casks

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Risk Task Group

October 20, 2003
Nuclear Safety Research Conference

Background

- Risk Task Group (RTG) is focal point for NMSS risk informing initiatives
- RTG is responsible for assisting NMSS Division with their risk-informing initiatives
- Risk-Informing tools
  - Risk-Informed Decision-Making (RIDM) Process
    - Screening Considerations
    - RIDM Decision-Making Algorithms
    - Risk Guidelines
- Pilot Studies

Pilot Study Objectives

- Identify modifications to RIDM guidance documents
  - Tested effectiveness of RIDM process and supporting guidance
  - Tested the logic of the RIDM decision-making algorithms
  - Tested draft NMSS risk guidelines
- Gain insights for future risk-informing activities in area of spent fuel storage

Pilot Study Topic

- Related to risk-informing guidance for conducting confinement reviews for casks
- RIDM process applied to issue previously implemented by staff in Interim Staff Guidance No. 18 (ISG-18)
- Issue: whether or not to modify acceptance criteria for conducting leakage tests and dose calculation associated with hypothetical release

Approach

- Using draft guidance, applied RIDM process
  - Step 1 - Define proposed action and alternatives
  - Step 2 - Apply screening considerations
    - Identify potential benefits
    - Assess feasibility
  - Step 3 – Evaluate risk information
  - Step 4 – Decide whether to implement proposed action

Step 1 - Definition of Proposed Regulatory Action

- Proposed Regulatory Action
  - Remove requirements for leakage testing and hypothetical off-site dose calculations and modify existing staff guidance for conducting confinement reviews of certain all-welded spent fuel canisters
- Considered various alternatives
  - Option 1 – proposed action
  - Option 2 – pre-ISG-18 approach
  - Others

Step 2 – Application of Screening Considerations

- Benefit
  - Help resolve a safety question?
  - Improve efficiency or effectiveness?
  - Reduce unnecessary regulatory burden?
  - Help effectively communicate reg decision?

Response was YES to at least 1 SC

- Feasibility
  - Availability of quality information?
  - Cost-effective to risk-inform?
  - Other factors that limit use of risk-informed approach?

Response was favorable for each SC

- **Proposed regulatory action was screened-in**

Step 3 – Evaluation of Risk Information

- Calculated individual accident risks for Option 1 and Option 2
  - Leakage was accounted for; doses extrapolated
  - Identified populations at most risk
  - Estimated facilities realistically affected
  - Best estimates staff judgment used for many input values
  - <u>Assumed</u> uncertainties in risk estimates were 2 orders of magnitude

Step 4 – Evaluate Decision: Risk Information

- Very small increase in risk to the public and workers
- Total individual accident risks estimated to be insignificant
- Storage cask performance/safety record gives a sense that overall risks of dry cask storage are low
- From a risk perspective, the proposed action should be allowed

Step 4 – Evaluate Decision: Other Information

- Maintain many layers of defense-in-depth
- Adequate margins of safety are maintained
- Net benefit (positive $) estimated
- This information suggests that the proposed action should be allowed

Conclusions

- Major outcomes of pilot study
  - Proposed action should be allowed
  - Conclusion consistent with the staff's earlier decision to implement ISG-18
  - Working group identified modifications to RIDM guidance

United States
Nuclear Regulatory Commission

FIRE RISK RESEARCH PROGRAM: ADVANCES SUPPORTING RISK-INFORMED REGULATION

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Office of Nuclear Regulatory Research

Presented at
Nuclear Safety Research Conference
Washington, D.C. • 20th October 2003

REGULATORY ACTIVITIES

- **Inspection**
  - **Associated circuits**
- **Reactor Oversight Process**
  - **Fire protection SDP revision**
- **Rulemaking**
  - **Risk-informed, performance-based fire protection rulemaking (endorsing NFPA 805)**

ASSOCIATED CIRCUITS

- **Issue is spurious operation of equipment**
- **Previous condition: All cable/circuit features candidates for inspection**
- **Scope of features narrowed by probability for resumption of inspection**
  - **Intra-cable**
  - **Thermoplastic inter-cable**
- **Significance of other features to be examined**
- **If testing needed, FIDECC is potential source**
- **Challenge: determine number of spurious actuations for risk analysis**
- **Circuit analysis report: NUREG/CR-6834**

FIRE PROTECTION SDP REVISION

- **Determines significance of fire protection inspection findings**
- **Revision more consistent with FRA framework**
- **Advances in areas include:**
  - **Challenging fires**
  - **Scenario fire bins**
  - **Low probability, potentially high consequence fires**
  - **Detection/suppression**

FIRE PROTECTION RULEMAKING (NFPA 805)

- **Guidance for reviews**
  - **Fire model**
    - ✓ **V&V: Five Rev. 1, FDTs, CFAST, FDS**
    - ✓ **Utilize ASTM standard 1355-97**
  - **Inputs to fire models, e.g. heat release rates**
  - **FRA methods, tools, data, including**
    - ✓ **Circuit analysis, HRA, plant model**
- **Understanding analysis supports development of review guidance**
- **ANS full power fire risk standard eventually forms basis**

ANTICIPATORY RESEARCH PROVIDES FOUNDATION

- **Fire modeling benchmark/validation**
  - **Multi-national blind benchmark exercises**
    - ✓ **Cable tray fires – model predictions similar (NUREG-1758)**
    - ✓ **Turbine hall fires – challenges predictive capabilities of models; e.g. vertical flow through hatches/gratings**
  - **Tests successfully completed at NIST, data being analyzed/compared to blind predictions (see poster session for test description)**
  - **Multi-compartment tests planned at DIVA (IRSN)**

ANTICIPATORY RESEARCH PROVIDES FOUNDATION (cont.)

- **Fire risk requantification study**
  - **Joint NRC/EPRI program on methods, tools, and data**
  - **Assessing feasibility of low power and shutdown study**

SUMMARY

- **Fire risk research program continuing to provide critical support to regulatory activities for nuclear power plants**
- **Anticipatory research provides foundation**
  - **Joint NRC/EPRI fire risk requantification program**
  - **Fire modeling benchmark/validation**
- **Risk considerations lead to more realistic safety decisions**
  - **Focus inspections in associated circuits**
  - **Provide improved technical methods for evaluating the significance of fire protection inspection findings**
  - **Allow for integrated safety analysis of plant changes under fire protection rulemaking (NFPA 805)**

BACKGROUND

QUANTIFICATION APPROACH

- **Overall quantification approach has not changed from earliest commercial NPP FRAs**
- **Approach remains:**

$$CDF = \sum_i \lambda_i \left(\sum_j p_{ed,j|i} \left(\sum_k p_{CD,k|i,j} \right) \right)$$

where λ_i = fire frequency for scenario i
p_{ed} = probability of equipment damage j due to fire
p_{CD} = probability of core damage from sequences k due to fire

OPTION 3:
RISK-INFORMED ALTERNATIV
10 CFR 50.46/GDC 35

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Nuclear Safety Research Conferen
Marriot at Metro Center
Washington, DC
October 20, 2003

SECY PAPER HISTORY

- SECY-98-300, December 1998, proposed high lev
for risk-informing Part 50
- SECY-99-264, November 1999, described the ove
plan for Option 3
- SECY-00-0198, September 2000, described chang
recommended for 50.44 and a proposed framewor
considering other changes to Part 50

SECY PAPER HISTORY (cont.)

- SECY-01-0133, July 2001, provided preliminary fea studies recommending risk-informed changes to Pa

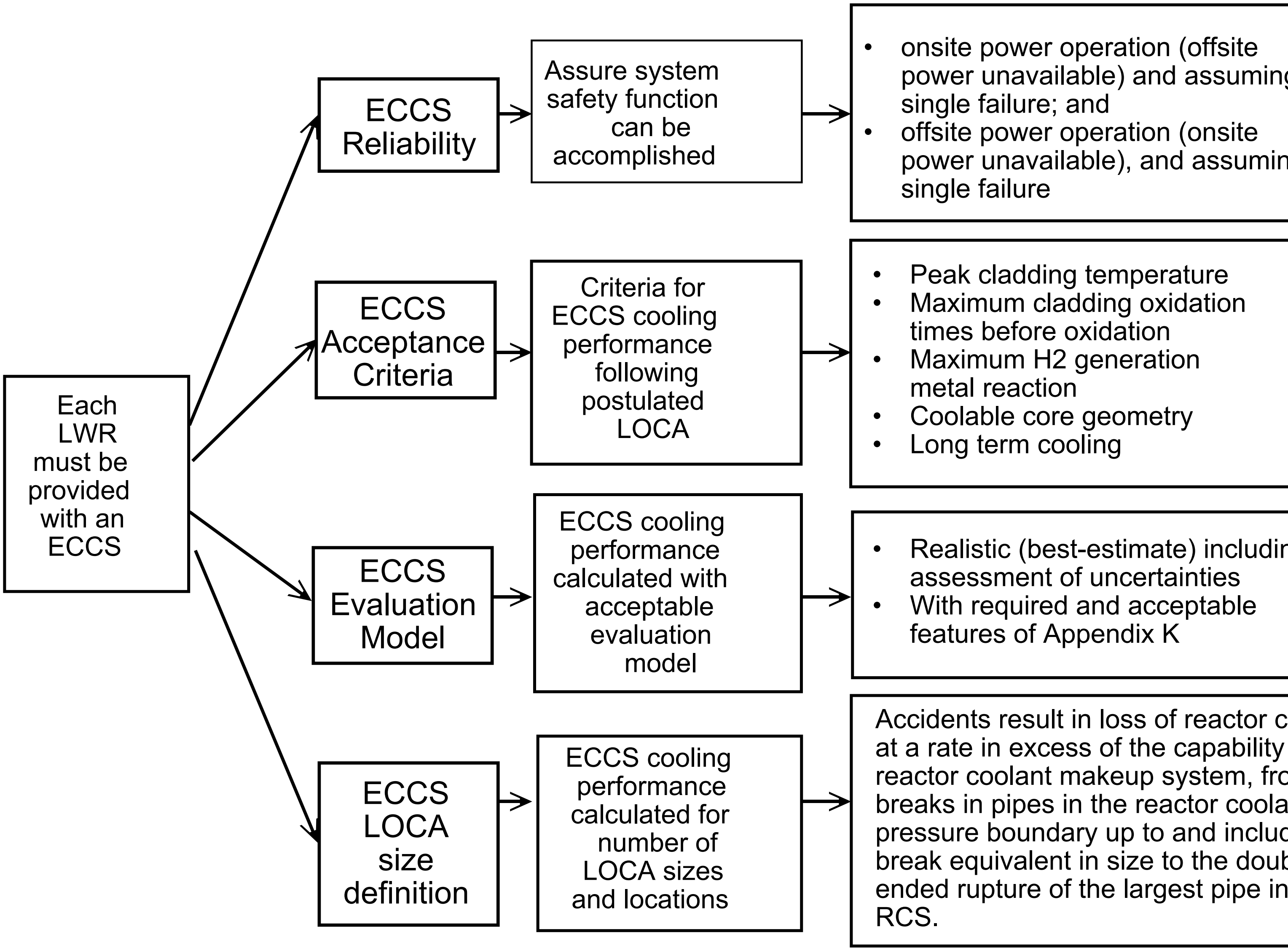
- SECY-02-0057, March 2002, provided recommend changes to 50.46 in the areas of:
  - ECCS break size redefinition
  - ECCS acceptance criteria
  - ECCS reliability
  - ECCS evaluation model

PUBLIC MEETINGS AND INTERACTIONS

- June 9, 2003 and July 24, 2003--Industry noted key SRM with which they might have difficulty, e.g., sco allowed changes, PRA scope and use of best-estim models

- September 5, 2003--Draft NEI white paper on Optic LOCA redefinition

OVERVIEW OF 50.46 (including Appendix K and GDC 35)



COMMISSION DIRECTIVES

- **Break size redefinition**
- Provide a comprehensive “LOCA failure analysis and estimation”
- Provide a proposed rule that allows for a risk informed alternative to the present maximum LOCA break size

- **ECCS acceptance criteria**
- Provide performance-based acceptance criteria for fue integrity, maintenance of core coolable geometry and core cooling

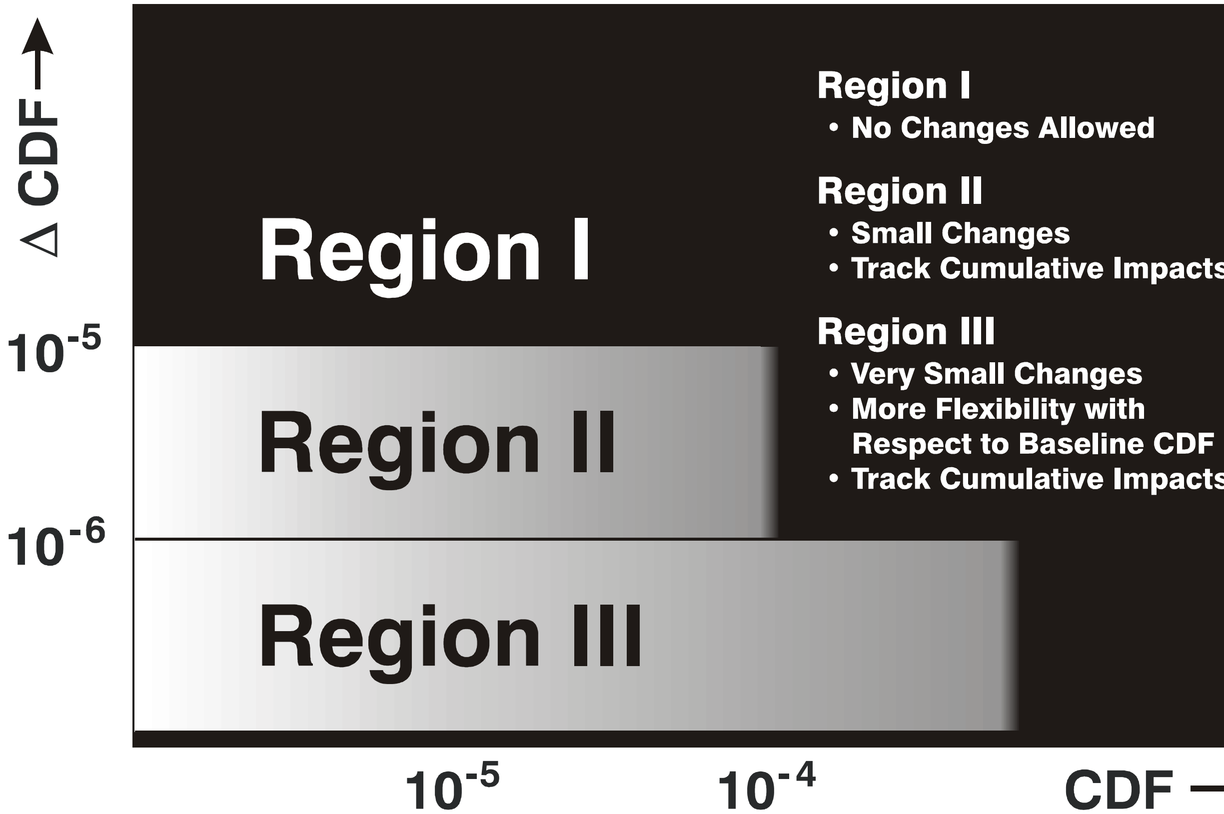
- **ECCS reliability**
- As an option, replace LOCA/LOOP requirements with ECCS reliability requirement that is communsurate wi LOCA frequency
- Include the need for a high quality PRA

- **ECCS evaluation model**
- Any changes that redefine the design basis LB LOCA use best estimate codes

CANDIDATE RISK METRICS

- CDF and LERF appear to be most likely me
- RG 1.174 acceptance guidelines may be us
  limit risk increase
- Defense-in-depth principles should be mair

Reg Guide 1.174 review



LARGE BREAK LOCA REDEFINIT

Objectives:

- Provide a comprehensive LOCA failure analysis an frequency estimation
- Prepare a proposed risk-informed rule allowing for alternative maximum LOCA break size
- Complete by March 2004

LOCA REDEFINITION GROUNDR

- Voluntary Risk-Informed Alternative which establish cutoff for break sizes to be removed from the desigr
- Any proposed functional changes should be risk-info and consistent with the principles of RG1.174
- ECCS functional reliability should be commensurate frequency of accidents in which ECCS success wou prevent core damage or a large early release.
- No changes to functional requirements unless fully r informed (For example, no change to ECCS coolant rates or containment capabilities to mitigate acciden
- Only the non-significant contributions to risk should handled through severe risk accident management

LOCA REDEFINITION GROUNDR (CON'T.)

- Realistically conservative estimates, with appropriat for uncertainly
- Full scope, high quality Level 2 PRA with internal an external initiators and all modes of operation, subjec review and NRC endorsement
- Use a 10-year period for the estimation of LOCA fre redistributions, with re-estimation every 10 years an of new type of failures every 5 years.
- Operational changes should be reversible if future re estimation of risk results in unacceptable frequency increases

ISSUES TO RESOLVE PRIOR TO RULE ISSUANCE

- Definition of new maximum break size metric that w
for the establishment of a new maximum break (e.g
frequency)
- Definition of plant change acceptance criteria
- Definition of licensee submittal requirements and st
review
- Definition of need for ongoing reversibility and mon
- Extent of current requirements to be removed

CURRENT STAFF THOUGHTS

- Staff and NEI appear to agree that some mitigation capability needs to be provided for breaks that are of the design basis but would be removed under th rule

- No definitive proposal has been established as to w mitigative capability should be

- Possible mitigative criteria might be no vessel failu shown by a realistic thermal hydraulic analysis or c deterministic criteria

- Proposed plant changes should also be included ir at the conclusion that the calculated core melt and containment failure frequencies remain lowt

ECCS ACCEPTANCE CRITERIA

Objectives:

- Proceed with the development of an optional perfor based approach to meeting ECCS acceptance crite
- New requirement might be designed to:
  - demonstrate adequate post-quench cladding ductility
- Allow use of cladding materials other than Zircaloy without licensees having to submit an exemption re
- Licensees expected to provide an adequate basis t that the new performance based criteria are met

ECCS EVALUATION MODEL

March 2003 SRM states:

- The Commision disapproved the staff's proposal to a voluntary alternative to Appendix K which would r the 1971 ANS decay curve with the 1994 standard
- Licensees are encouraged to use best estimate eva model codes when evaluating the adequacy of cha made to the plant to insure conformance with accep criteria

ECCS RELIABILITY

Objectives:

- Commission approved the staff's recommendation proposing rulemaking for a new, voluntary provisio inform GDC 35 in the area of LB LOCA and coincic LOOP
- Continued stakeholder input should be sought
- Other issues, such as delayed LOOP, should be co
- Commission also agreed to allow the staff to study change to the single failure criterion

CONDITIONAL PROBABILITY OF LOOP GIVEN LOCA

- Dependence between LOCA (or any reactor t a consequential LOOP

- Extremely limited data for LOOP given LOCA

- Plant-specific method for assessing probabilit LOOP given LOCA is under consideration
  - Transient (grid-related) factors
  - Plant-centered factors (failures of plant electrical eq

WORK REQUIRED TO SUPPORT LOOP/LOCA RULE CHANGE

- Determine appropriate reliability and CDF threshol
- Identify features that tend to impact the likelihood o offsite power following a LOCA
- Determine acceptable methods and assumptions fo estimating plant-specific probability of loss of offsite given a LOCA.
- Support development of the regulatory guides need implementing the recommended risk-informed alte rule

THE SINGLE FAILURE CRITERIO

- GDC 35 requires that the ECCS safety function be accomplished assuming a single failure

- Staff will evaluate the basis for revising or replacin single failure criterion in the alternative rule, but or affects ECCS

- The single failure criterion is applied to more than ECCS. GDCs 17, 34, 38, 41 and 44 also contain t failure criterion.

- Reg Guide 1.53, "Application of Single-Failure Cri Safety Systems," discusses the issue in relation to systems

- Research findings due July 2004

STAFF PLANS

- Provide Commission with staff's response to the SR
discussing the difficulties associated with: PRA sco
reversibility, defense in depth, applicability to future
plant monitoring, extent of plant changes envisione
- Conduct additional public meetings
- Complete preliminary estimates of pipe break frequ
across break sizes and causes within the next few
- Develop additional technical bases for the rule

Redefinition of LOCA Break Size and Frequency

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U.S. Nuclear Regulatory Commission

Nuclear Safety Research Conference
Washington, DC
October 20, 2003

Motivation for Elicitation

- NRC is developing risk-informed changes to the ECCS acceptance criteria within 10 CFR 50.46 in the following technical areas:
  - **ECCS Reliability: Demonstrate safety without assuming LOOP coincident with single point failure in the design basis**.
  - ECCS Acceptance Criteria
  - ECCS Evaluation Model
  - **LB LOCA size redefinition and location**.
- The frequencies of LOCA initiating events is fundamental to the assessment of ECCS reliability and the LB LOCA size redefinition.

Historical LOCA Frequencies Evaluation

- LOCA frequencies previously based solely on operating history.
- Notable Previous Evaluations:
  - WASH-1400 (1975): Estimates largely based on experience in other industries due to lack of reactor experience.
  - NUREG-1150 (1987): Updated the WASH-1400 distributions to account for the additional service since WASH-1400.
  - NUREG/CR-5750, Appendix J (1998): Small break frequencies were updated from the original WASH-1400 study while the medium and large break frequencies were calculated from precursor leaks in class 1 systems.
  - Barsebäck-1 Study (1998): Determined estimates using piping reliability attribute and influence characteristics for each distinct degradation mechanism.

LOCA Frequency Reevaluation: Limitations of Operating Experience

- Comprehensive database required to accurately assess importance of rare events (No LBLOCA in LWR operating history).
  - Reporting requirements and accuracy are variable and depend on the degradation mechanism and piping system.
  - Passive system failures are reported using various mechanisms.
- Not necessarily representative of future system performance.
  - Material aging and environmental effects are not always accurately captured in the experience base.
  - No similar maintenance plan as with active components to ensure applicability historical failure rates.
- Methodology based on existence of precursor event prior to failure.
  - Not all degradation mechanisms exhibit a precursor leak prior to failure.
  - Development of the conditional failure probability given a precursor event is mechanism specific and has historically been poorly known.

LOCA Frequency Reevaluation: Technical Issues

- LOCA contribution variables span broad technical fields
  - PFM, piping design, piping fabrication, operating experience, materials, degradation mechanisms, thermo hydraulics, operating mitigation practices, stress analysis, nondestructive evaluation, human factors, etc.
  - Integrated, multidisciplinary approach required.
- Impact of material aging and environmental effects must be considered and benchmarked against operating experience.
- Contribution of both leaking and non-leaking LOCA contributors must be considered.
  - Credit leak detection (leak-before-break) as appropriate.
  - Identify and quantify contributions of non-leaking cracks prior to failure.
- LOCA estimates must consider uncertainties.

LOCA Frequency Reevaluation: Expert Elicitation Process

- **Expert opinion (elicitation) is a formal process for providing quantitative estimates for the frequency of physical phenomena when the required data is sparse or when the subject is too complex to adequately model.**
  - The rarity of LOCA events (data sparseness) is evident.
  - Complexity is evident in the enormous pipe system variables which must be considered to model from first principles.
  - Complexity also exists due to the many potential non-pipe LOCA failure modes which contribute to the LOCA spectrum.
- **Elicitation has been used successfully to solve similar problems.**
  - Development of seismic hazard curves.
  - Performance assessments for high-level radioactive waste repository.
  - Determination of reactor pressure vessel flaw distributions.

Elicitation Scope and Objectives

- Develop piping and non-piping passive system LOCA frequencies as a function of leak rate and operating time up to the end of the license extension period.
- Determine LOCA frequency distributions for typical plant operational cycle and history.
- Estimate conditional LOCA probability distributions for rarer, emergency faulted load conditions.
  - Seismic loading.
  - Other large, unexpected internal and external loads.

Formal Elicitation Approach

- **Select panel and facilitation team.**
- Develop technical issues.
- Quantify base case estimates.
  - Develop quantitative estimates for well-defined piping conditions.
  - Two estimates using PFM and two estimates from service history analysis.
- Formulate elicitation questions.
- Conduct individual elicitations.
- Analyze quantitative results and qualitative rationale.
- Summarize and document results.

LOCA Frequency Reevaluation: Expert Elicitation Approach

- **Pilot Elicitation (Complete).**
  - Conducted last year using 11 internal (NRC) experts with broad knowledge-base.
  - Provided interim results for evaluation of ECCS reliability.
  - Developed possible framework for subsequent elicitation and identified strengths and weaknesses to address in formal elicitation.
  - Identified technical issues for consideration within formal elicitation.
- **Formal Elicitation (Ongoing).**
  - Individual elicitations conducted for each expert that is monitored by a facilitation team.
  - Twelve external experts assembled from nuclear industry, DOE laboratories, consultants, and international regulatory agencies with broad knowledge-base.
  - Facilitation team is comprised largely of NRC personnel.

Formal Elicitation Approach

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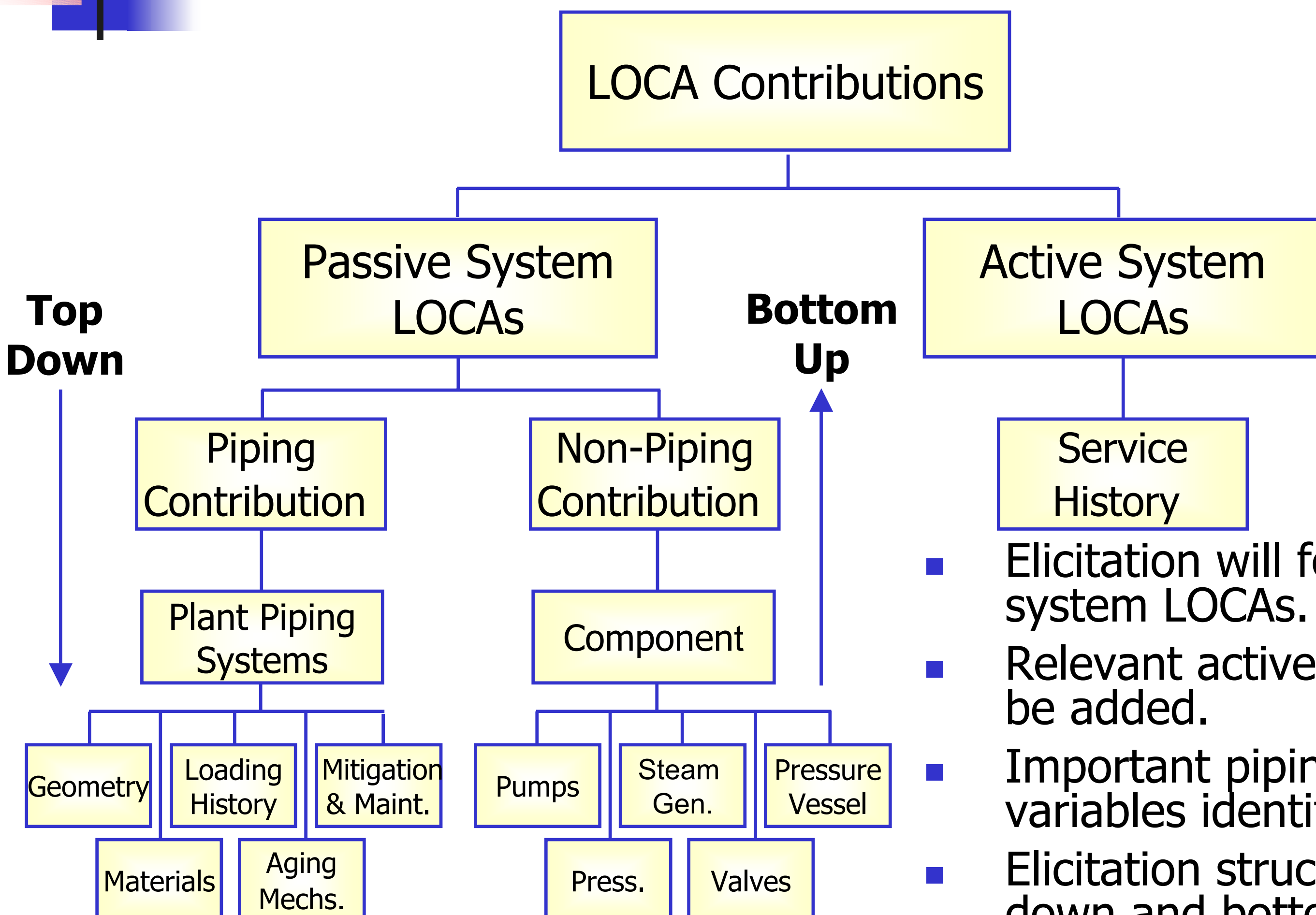
LOCA Sizes and Operating Time Periods Evaluated

- LOCA sizes based on leak rate to group plant system response characteristics.
- First three categories encompassed traditional definitions utilized in NUREG-1150 and NUREG/CR-5750.
- Three more LBLOCA categories added to examine trends with larger break sizes.
- Correlation between leak rate and break size developed for relevant BWR and PWR systems.

Category	Leak Rate Threshold (gpm)	LOCA Size
1	> 100	SB
2	> 1500	MB
3	> 5000	LB
4	> 25,000	LB a
5	> 100,000	LB b
6	> 500,000	LB c

- Three time periods evaluated.
  - Current (average 25 years of operating experience.
  - End of design life (40 years of operation).
  - End of life extension (60 years of operation).

General Issue Classification



- Elicitation will focus on passive system LOCAs.
- Relevant active system LOCAs will be added.
- Important piping and non-piping variables identified.
- Elicitation structure will support top down and bottom up analysis.

Formal Elicitation Approach

- Select panel and facilitation team.
- Develop technical issues.
- **Quantify base case estimates.**
  - **Develop quantitative estimates for well-defined piping conditions.**
  - **Two estimates using PFM and two estimates from service history analysis.**
- Formulate elicitation questions.
- Conduct individual elicitations.
- Analyze quantitative results and qualitative rationale.
- Summarize and document results.

Piping Base Case Development

- The base cases will be used to anchor the elicitation responses.
- Base case conditions specify the piping system, piping size, material, loading, degradation mechanism(s), and mitigation procedures.
- Five Base Cases Defined.
  - BWR
    - Recirculation System
    - Feedwater System
  - PWR
    - Hot Leg
    - Surge Line
    - High Pressure Injection makeup.
- The LOCA frequency contribution (per year) of each set of base case conditions will be calculated as a function of leak rate and operating time.
- Four panel members chosen to perform calculations: two using operating experience and two using probabilistic fracture mechanics.

Non-Piping Base Case Development

- The non-piping base cases could have been developed in a similar manner to the piping base cases.
  - Choose several representative systems.
  - Examine and extrapolate operating experience through modeling
- However, the variety and complexity of the non-piping failure mechanisms makes this assessment intractable and of limited value.
- Philosophy here is to conduct database searches for each non-piping failure mechanism listed to develop leaking component frequencies.
- These frequencies will be used to anchor the non-piping responses for each expert.
- Each expert must determine how to translate the leaking and crack frequency information into meaningful LOCA estimates.

Formal Elicitation Approach

- Select panel and facilitation team.
- Develop technical issues.
  - Define scope and objectives of elicitation.
  - Construct approach for determining LOCA frequencies.
  - Determine significant issues affecting LOCA frequencies.
- Quantify base case estimates.
  - Develop quantitative estimates for well-defined piping conditions.
  - Two estimates using PFM and two estimates from service history analysis.
- **Formulate elicitation questions.**
- Conduct individual elicitations.
- Analyze quantitative results and qualitative rationale.
- Summarize and document results.

Elicitation Question Development

- Questions focus on the following topic areas.
  - Base Case Evaluation.
  - Regulatory and Utility Safety Culture pertaining to LOCA initiating events.
  - LOCA frequencies of Piping Components.
  - LOCA frequencies of Non-Piping Components.
  - Conditional piping failure under Emergency Faulted Loading.
  - Conditional non-piping failure under Emergency Faulted Loading.
- Questions are asked relevant to a set of conditions and quantitatively linked to the base case results.
- Each question asks for mid, low, and high values for each question as well as appropriate rationale or comments.
- Questions can be answered using a top-down or bottom-up approach.
- Rationale is discussed for important issues provided by each expert.

Ongoing and Future Elicitation Work

- Conduct individual elicitations.
  - Provide answers to questions and rationale for answers.
  - Discuss significant issues which impact LOCA frequency estimation.
  - Elicitations completion date is targeting early November.
- Analyze quantitative results and qualitative rationale.
  - Calculate results for each expert if appropriate.
  - Combine answers for individual questions and calculate results.
  - Propagate uncertainties.
- Conduct wrap-up meeting.
  - Summarize quantitative and qualitative results.
  - Summarize analysis methodology and LOCA results.
  - Obtain feedback from the expert panel.
- Summarize and document results.

Summary

- NRC is using expert elicitation process to develop LOCA initiating event frequency distributions as a function of effective break size.
- The results will be used as a technical basis for 10 CFR 50.46 Option III revision.
- Elicitation process is designed to capture uncertainties expressed by wide-ranging technical opinions to a complex topic area where the underlying data is sparse.
- The process has developed quantitative estimates of simplified conditions used to anchor subsequent responses (base cases) that are based on extrapolations of operating experience.
- Experts must determine relevant issues/parameters which govern LOCA frequency estimates and provide the relative between these issues/parameters and the set of anchor conditions.

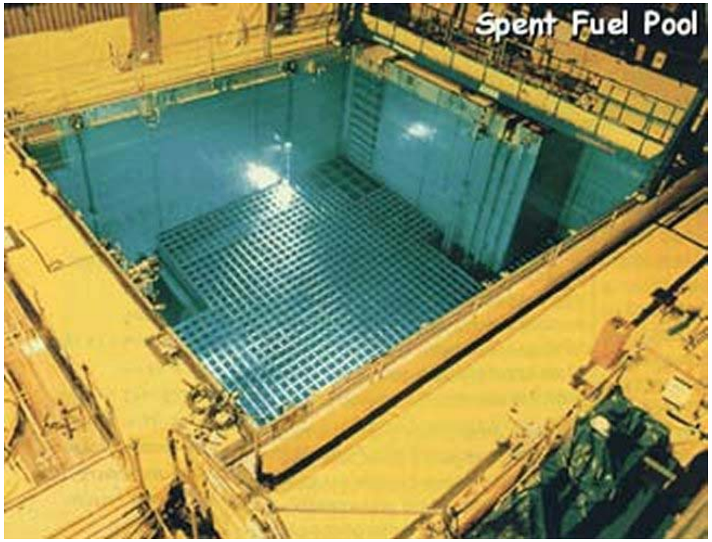
More Realistic Assessment of Spent Fuel Pool Accidents

October 20, 2003

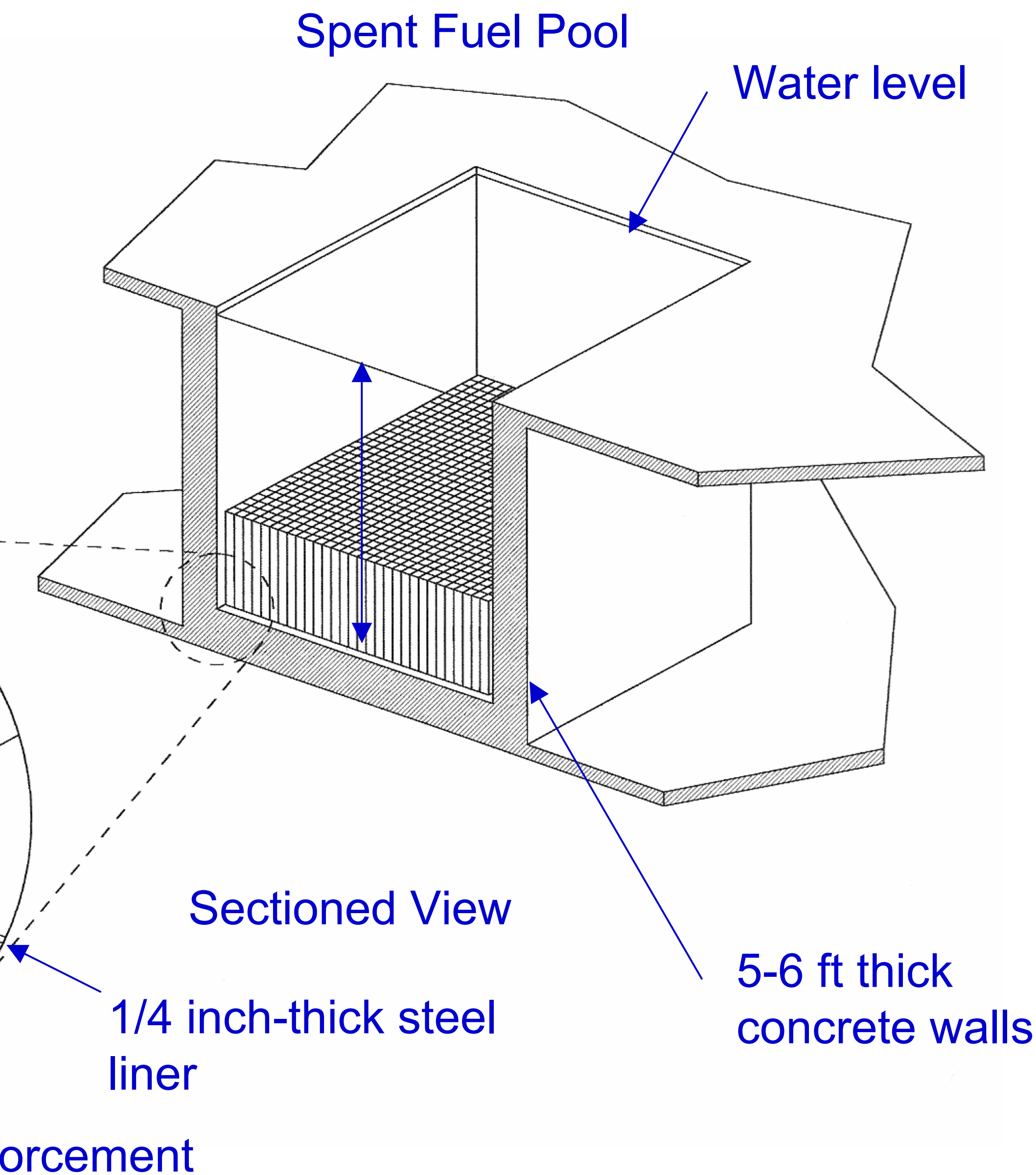
Jason H. Schaperow
Office Of Nuclear Regulatory Research
U.S. NRC

Spent Fuel Safety

- **Staff concludes that public health and safety is protected with spent fuel stored in pools or dry casks**
- **Spent fuel pools are robust structures constructed of reinforced, thick concrete walls with stainless steel liners. Pools may be further protected by surrounding structures or located underground**



Spent Fuel Pool



Spent Fuel Pool
Water level
Sectioned View
1/4 inch-thick steel liner
5-6 ft thick concrete walls
Reinforcement

Spent Fuel Pool - Background

- The fuel in the spent fuel pool generates small fraction of the heat in the reactor
  – Fuel in spent fuel pool which is relatively full (e.g., containing 4 reactor cores) generates heat at a rate which is 10 to 40 times lower than that of fuel in reactor when reactor is shutdown
  – Lower heat generating capacity of spent fuel means heat removal is simple, even under adverse conditions
- Most of the heat generated by fuel in the spent fuel pool comes from the fuel most recently offloaded from the reactor – not the old fuel which may be loaded in casks

Spent Fuel Pool Studies

- Past NRC studies of spent fuel pools have used very conservative models/methods and assumptions to evaluate potential for fuel heatup, fission product release (radiation) and offsite consequences
  - Bounding pool conditions
  - Simplified/conservative models for fuel heatup
  - Limited or no credit for fission product release attenuation

Spent Fuel Pool Studies

- Very conservative analyses were adequate for the original intended purpose where more realistic and accurate (and more detailed) evaluation was not needed
- When past studies are taken out of original context, where applied to very low probability events, the predicted behavior including consequences are not appropriate
  - Risk = Frequency x Consequences

More Realistic SFP Analysis

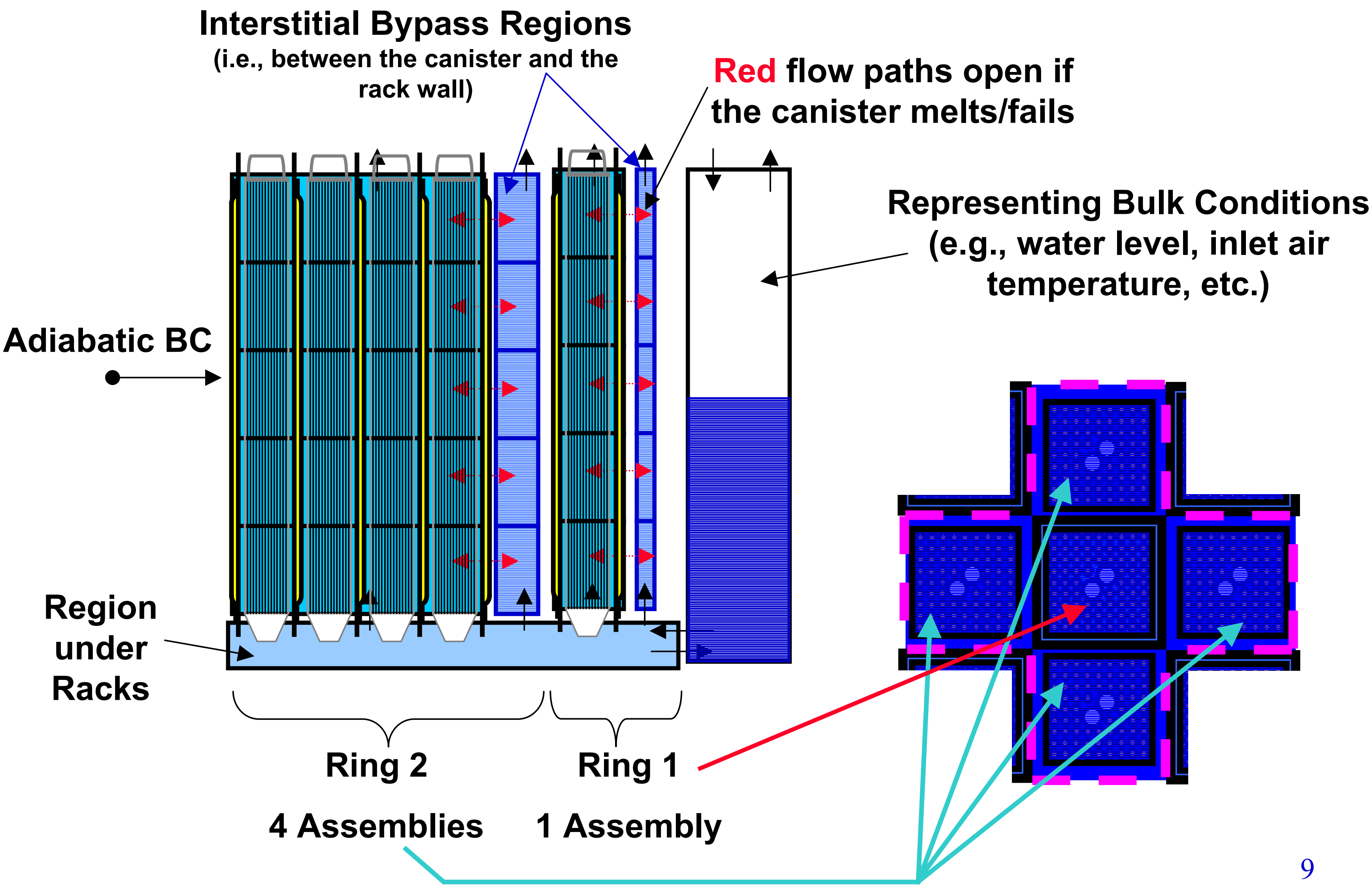
- NRC Vulnerability Project
  - Past work primarily limited to “early phase” heat-up calculations, no integrated severe accident analysis performed
  - Most codes only analyzed potential for zirconium fire using “ignition temp” criteria

    - No Severe Accident Models
    - Historical Tools Also Criticized for Modeling Limitations
      - Damage propagation
      - Oxidant depletion
      - FP release and transport modeling
      - Heat transfer modeling
      - Flow Mixing
    - Shortcomings can be overcome with state-of-the-art severe accident modeling

Modeling Approach

- 2 Model Approach - Separate Effects and Whole Pool/Reactor Building Models
- Separate Effects Model
  - Developed First to Guide Full SFP Model Development
  - Fast Running + Controlled Boundary Conditions
  - Accurately Represents Single Assembly Geometry + 4 Neighbors
  - Some Modeling Issues Being Resolved
    - Use Separate Effects Model to Develop Appropriate Modeling Approach
    - Identify Sensitivities and Uncertainties
    - Recommend Code Development
- Full SFP + Building Model
  - Integral Effects
  - Whole SFP Source Term

Separate Effects SFP Model



Status

- More detailed modeling and analysis is underway
  - Based on actual pool conditions, fuel inventory and loading pattern
- Insights from ongoing analyses indicate that fuel in the spent fuel pool may be much more easily cooled than predicted in earlier studies
- Ongoing analyses also indicate that even if cooling is lost more time is available to restore cooling and prevent fuel damage
- Ongoing analyses indicates that even if fuel is damaged consequences will be reduced from past studies
- Ongoing work is evaluating effectiveness of potential mitigative options for enhancing the coolability of spent fuel in pool storage